UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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August 10, 2000

Craig Anderson, Vice President
Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, Arkansas 72801-0967

SUBJECT: NRC's ARKANSAS NUCLEAR ONE INSPECTION REPORT NO. 50-313/00-14; 50-368/00-14

Dear Mr. Anderson:

This refers to the inspection conducted on July 27 to August 8, 2000, at the Arkansas Nuclear One, Unit 2 facility. The enclosed report presents the results of this inspection. A technical debrief was discussed on July 28, 2000, with Mr. M. Smith and other members of your staff. The results of this inspection were discussed on August 8, 2000, with Mr. J. Vandergrift and other members of your staff.

The inspection was an examination of activities conducted under your licenses as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your licenses. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on reactor safety.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

P. Harrell, Chief
Project Branch D
Division of Reactor Projects

Docket No.: 50-313
50-368
License No.: NPR-51
NPF-6

Enclosure:
NRC Inspection Report No.
50-313/00-14; 50-368/00-14

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Dale Thatcher **(DFT)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-313; 50-368
License No.:	NPR-51; NPF-6
Report No.:	50-313/00-14; 50-368/00-14
Licensee:	Entergy Operations, Inc.
Facility:	Arkansas Nuclear One, Units 1 and 2
Location:	1448 S. R. 333 Russellville, Arkansas 72801
Dates:	July 27 through August 8, 2000
Inspectors:	R. Bywater, Senior Resident Inspector C. Beardslee, Materials Engineer, Office of Nuclear Reactor Regulation
Accompanying Personnel:	T. Alexion, Project Manager, Office of Nuclear Reactor Regulation
Approved by:	P. Harrell, Chief, Project Branch D Division of Reactor Projects

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Arkansas Nuclear One
NRC Inspection Report 50-313/00-14; 50-368/2000-14

This report covers onsite inspection and in-office review of Unit 2 steam generator inservice inspection surveillance activities. In the Reactor Safety area, the cornerstones inspected included Mitigating Systems and Barrier Integrity.

There were no inspection findings identified in these areas.

Report Details

Summary of Plant Status

Unit 2 was shutdown for steam generator inspection (Outage 2P00-1) during this inspection.

1. REACTOR SAFETY
Cornerstones: Mitigating Systems, Barrier Integrity

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's program for monitoring steam generator tube degradation. As part of this review the inspectors:

- Reviewed documents defining the scope of eddy current testing planned for steam generator tube examinations, including number of tubes, locations, scope expansion criteria, and specific eddy current probes.

- Reviewed the eddy current data collection, management and analysis methods.

- Observed a sample of in-progress eddy current data collection.

- Reviewed the licensee's selection criteria for determining which steam generator tubes were to be in-situ pressure tested.

- Reviewed the eddy current inspection results and the licensee's selection of tubes to be in-situ pressure tested.

- Reviewed the licensee's criteria for determining which steam generator tubes were to be plugged.

b. Findings

The licensee performed eddy current inspections of the Unit 2 steam generator tubes to determine which tubes were defective and required repair or plugging. In addition, the licensee performed in-situ pressure testing of a small number of defective tubes to evaluate their structural and leakage integrity.

The eddy current inspection scope consisted of bobbin coil probe inspections of all inservice tubes, and rotating pancake coil probe inspections of all indications identified by the bobbin coil inspection. The inspectors determined that the planned inspection scope was in accordance with the Technical Specifications and was being appropriately controlled by the eddy current data management group. In addition, the observed data collection, management and analysis methods were in accordance with plant procedures. The licensee indicated that the inspection equipment, data collection procedures, and data analysis techniques being used during this inspection were essentially the same as those used during the previous steam generator inspection (November 1999). The inspectors did not identify any activities contrary to this assertion, and therefore would expect the sensitivity of the Outage 2P00-1 inspection to be

similar to the November 1999 inspection.

All flaws identified during the Outage 2P00-1 inspection were axial outside diameter stress corrosion cracking at the eggcrate supports. The licensee identified 64 flaws in 58 tubes in Steam Generator A and 148 flaws in 131 tubes in Steam Generator B. These results were within the range projected by the licensee.

The licensee performed in-situ pressure testing on a small number of defective steam generator tubes to evaluate their structural and leakage integrity. They developed selection criteria for determining which steam generator tubes were to be in-situ pressure tested. The selection criteria were documented in Revision 1 of Engineering Report ER-974855-E205, and consisted of a combination of estimated flaw length, maximum depth, and average depth. The inspectors reviewed the in-situ selection criteria as well as the licensee's selection of tubes to be in-situ pressure tested.

One tube in Steam Generator A and seven tubes in Steam Generator B were selected to be pressure tested for leakage integrity at main steam line break (MSLB) pressure differentials. One Steam Generator A tube and three Steam Generator B tubes leaked a minimal amount at MSLB conditions. Five tubes were selected to be pressure tested for structural integrity at three times the normal operating pressure differential (3dp), four of which had leaked at MSLB conditions. Four of the five tubes were tested to approximately 500 psi above 3dp with no failure. The licensee indicated that the fifth tube (Tube 40/108) passed the 3dp pressure, but estimated that it burst at less than 100 psi above 3dp. Tube 40/108 had not met the selection criteria for pressure testing at 3dp (the flaw was estimated to be too short to burst), but the licensee elected to perform this test because the tube leaked when tested at MSLB pressure differentials.

Based on the 3dp pressure test results, the inspectors concluded that it appeared possible that this flaw might have burst at less than 3dp if the flaw had been deeper. This was a concern, because a flaw of this length would not have met the licensee's selection criteria for pressure testing regardless of its depth. Based on the inspector's concerns, the licensee reevaluated the selection criteria and determined that nondestructive evaluation uncertainties had not been appropriately considered when calculating the selection criteria. The selection criteria were modified and the licensee evaluated the inspection results to determine whether any additional tubes required in-situ pressure testing. The licensee concluded that no additional tubes required testing. The safety significance of this finding was considered very low based on the absence of adverse consequences. The inspectors determined that the appropriate tubes were in-situ pressure tested.

There were no significant findings identified during this inspection.

4. OTHER ACTIVITIES (OA)

4OA6 Management Meetings

.1 Exit Meeting Summary

A technical debrief was discussed on July 28, 2000, with Mr. M. Smith and other members of your staff. The results of this inspection were discussed on August 8, 2000, with Mr. J. Vandergrift and other members of your staff. The managers acknowledged the findings presented and also informed the inspectors that no proprietary material was examined during the inspection.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

S. Bennet, Licensing Specialist
M. Cooper, Licensing Specialist
M. Smith, Engineering Programs and Component Manager
D. Harrison, Engineering Programs Supervisor
J. Vandergrift, Director, Nuclear Safety

DOCUMENTS REVIEWED

Unit 2 Engineering Report ER-974855-E205	Steam Generator Pre-Outage Degradation Assessment and Repair Criteria for 2P00	Revisions 1
Unit 2 Training Manual ANO-2-OTH-ESP-SGMAN	Steam Generator Eddy Current Training Manual	Revision 4
Engineering Standard HES-28	ANO-2 Steam Generator Eddy Current Examination Guidelines	Revision 12
Procedure/Work Plan 5120.500	Steam Generator Integrity Program Implementation	Change 008-03-0
Procedure/Work Plan 5120.509	Steam Generator Inservice Inspection Implementation Plan	Change 001-00-0

LIST OF ACRONYMS AND INITIALS USED

MSLB - main steam line break
3dp - three times normal operating pressure differentials

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspections so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http:\\www.nrc.gov\NRR\OVERSIGHT\index.html